October 10, 1980 ANL FRA-TM-129

STATUS OF FAST REACTOR PHYSICS*

by

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**Work performed under the auspices of the U.S. Department of Energy.*

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ABSTRACT

A review of FBR physics is presented with an emphasis on the progress in recent years. The recent trends in FBR design (e.g., heterogeneous cores) and their impact on physics is discussed. Progress in the base program areas of nuclear data and methods and codes is reviewed and evaluated as are the critical experiment programs. Discussions are presented for many of the key FBR parameters. For some of these, such as sodium void and control rod worths, significant progress has been made in recent years. The ability of current reactor codes and nuclear data to predict the neutronic properties of FBR cores over a range of sizes, concepts, and configurations is reviewed. The currently outstanding physics discrepancies and deficiencies are discussed.

I. INTRODUCTION

In 1972 Avery[1] presented a review of fast breeder reactor physics at the Kiamesha Lake ANS topical meeting. Considerable progress has taken place in the intervening eight years although some old problems remain and new problems have emerged. In this paper the developments during this period, the work in progress and the outstanding problem areas and deficiencies are reviewed.

The initial section of this report discusses the developments and trends in fast reactor design and their impact on reactor physics. Such items as the emergence of the heterogeneous core concept, proliferation concerns that led to the short-lived interest in thorium and symbiotic fuel cycles, and the revived interest in fuels with better breeding potential (e.g., carbides) are covered. Concurrently, there have been significant developments in the physics base-program activities, some as the result of the continuation of ongoing activities and some in response to the design trends. These include the accumulation of a large and reliable fast integral experiment data base for prototypic cores, both homogeneous and heterogeneous, a significant increase in the power and the rigor of the reactor analysis codes, a significant increase in the complexity and the detail included in the nuclear data files, and advances in nuclear data measurements and evaluations. Discussions are included for each of these base program activities.

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This is followed by short discussions on many of the important areas of reactor physics and those where key developments have occurred in recent years. The list is not complete, and several areas (e.g., Doppler, gamma heating) could not be included due to space limitations. Also for this reason, only a minimum of references are listed. The emphasis in this paper is on the ANL physics program.

There has been a major effort in the Applied Physics Division at ANL over the past year to systematically reevaluate and recalculate, using consistent methods and data, the important information from the base program for several of the key physics topics. This is called the Assessments Program. It was undertaken because of the extensive and reliable data base (especially from the criticals) now available. The objective is to document in detail the current state of the art for each topic along with recommended analysis prescriptions. The results revealed considerably improved consistency in many areas. The paragraphs on sodium void, control rods and criticality present some of the initial results.

II. PHYSICS CONSIDERATIONS AND TRENDS IN FAST REACTOR DESIGN

The fuel design for both FFTF and CRBR incorporates a small pin size: 0.23-in. diameter and 15-mil cladding thickness. A large number of mixed-oxide pins of such size have been irradiated in EBR-II in direct support of the FFTF and CRBR designs. However, from the neutronics viewpoint, the small pin size results in a poor breeding performance. Hence, the design studies of the last several years have emphasized the optimization of fuel design with respect to the breeding or, more precisely, the doubling time.

The breeding performance for fast breeder reactors is most strongly affected by fuel volume fraction, fuel type and fissile/fertile isotope ratio. As the fuel volume fraction is increased, for example by increasing the fuel pin diameter, the breeding ratio is improved because the hardened spectrum increases excess neutrons available and reduces parasitic absorption losses. However, the increased heavy metal loading also causes the fissile inventory to increase, hence the objective is to define an optimum fuel volume fraction which minimizes the doubling time.

For a range of feasible designs optimized for the doubling time, there is an inherent difference of 0.10-0.15 in the breeding ratio between the oxide and the carbide fuels and also between the carbide and the metal fuels. The breeding performance improvements for the carbide fuel over the oxide fuel, and for the metal fuel over the carbide fuel are traceable directly to the basic neutronic properties of each fuel type associated with the neutronic energy spectrum that is characteristic of each. About one-half of the breeding ratio improvement for advanced fuels is due to the increase in the η-value of fissile isotopes and the other half to the increase in the fertile fission bonus as the spectrum is hardened. Although advanced fuel development has been more actively pursued in recent years, the irradiation data base is still sparse. Hence, the mixed-oxide fuel remains as the reference FBR fuel at present, while some level of advanced fuels R&D work is continuing for a possible long-term application.

As part of the International Nuclear Fuel Cycle Evaluation (INFCE) and the Non-proliferation Alternative Systems Assessment Program (NASAP) studies, thorium-based alternative fast breeder cycles and symbiotic systems have been evaluated in recent years. The breeding performance and the uranium utilization of the overall system are dictated most strongly by the amount of thorium employed in the fast breeders. The breeding ratio is not affected much when thorium is utilized only in the external blankets in place of uranium because the blanket breeding is dictated by the core neutron leakage. However, the situation worsens when thorium is introduced into the core. Even when plutonium fueling is maintained, if thorium is used as the core fertile material, the breeding ratio penalty is substantial, due mainly to the reduction of the fertile fission bonus. When ^{233}U is substituted for plutonium in addition to the fertile substitution of thorium, the η-value is decreased and there is additional decrease in the fertile fissions due to the absence of ^{240}Pu and ^{242}Pu. Hence, the neutron economy and associated breeding characteristics of thorium-based fuel cycles are far inferior to those of the reference Pu/U cycle in the fast reactor. The safety-related physics parameters, sodium void reactivity and Doppler coefficient are improved for thorium-based cycles.

The most significant development in fast reactor design in recent years is the increased attention given to the heterogeneous core design concept. Although studied in the 1960s, the concept has not been given much attention until it was resurrected by the French in 1975 as a means for improving breeding performance. Since then it has been adopted in the CRBR design in order to improve the breeding ratio.

Numerous design studies have been carried out to quantify advantages and disadvantages of the heterogeneous concept relative to the homogeneous concept. The difference between the breeding performance of the two concepts is very small. In fact, the same conclusion extends to other core performance characteristics and fuel cycle economics. In most cases, the conclusions are dictated by the consistency in the comparison (optimized design for one concept versus non-optimized for the other, etc.) rather than by any inherent differences. In the case of CRBR, the breeding performance was improved by changing to the heterogeneous design because the fuel pin design was fixed. There are some inherent differences between the two concepts: for example, thermal striping of the upper internals may be worse in the heterogeneous design, but the fluence/burnup ratio is reduced. The most significant difference, though, is in the sodium-void reactivity characteristics. The heterogeneous core results in a lower core sodium-void reactivity and a greater incoherency in voiding, which provide advantages in the event of an hypothetical core disruptive accident (HCDA). This characteristic is the main rationale in the U.S. for considering the heterogeneous core configuration in large LMFBR designs. A heterogeneous arrangement has been adopted in the Conceptual Design Study (CDS) as the reference design.

The heterogeneous core concept has provided a driving force in the development of the fast reactor physics in recent years. The complexity of the heterogeneous configuration has demanded development of more rigorous and efficient computational tools in the areas of cross-section processing, neutronics, space-time kinetics, etc. In-depth analyses has been carried out to understand the physics of safety-related parameters as affected by various design changes, as well as the enhanced sensitivity of the flux distribution in heterogeneous configurations. Heterogeneous critical experiments have been carried out to identify unresolved physics issues and validate the analysis methodologies for demonstration size cores, but additional experiments are needed for large heterogeneous cores where spatial effects become more important.

III. CRITICAL EXPERIMENT PROGRAMS

Since 1970, ZPPR programs have encompassed two main activities -- design and licensing support for CRBR and benchmark studies of large homogeneous LMFBRs. The CRBR-related assemblies included benchmark and engineering-mockup studies of homogeneous cores in ZPPR-2 through -6 as well as heterogeneous cores in ZPPR-7, -8, and -11. From the standpoint of current design interest, it is the data from the heterogeneous cores and the large homogeneous cores, ZPPR-9 and -10, that provide the most relevant information. During the past six years, ZPR-9 has been used for special programs and to study alternative breeder concepts (e.g., GCFR, Carbide fuels, meltdown configurations).

A. LARGE HOMOGENEOUS FBR CRITICALS

The large homogeneous-core physics program, a cooperative U.S.-Japanese effort denoted as JUPITER, was based on critical-assembly simulations of 700-900 MW(e), two-zoned LMFBRs, with an optimized $(Pu/U)O_2$ fuel volume fraction of 0.4. The program started with ZPPR-9, a 4600ℓ cylindrical assembly which omitted such singularities as control rod positions. The ZPPR-10 program followed six months of measurements in the ZPPR-9 physics benchmark. Typical design features, including control rod positions and hexagonal zone boundaries, were a part of all ZPPR-10 cores. ZPPR-10A and -10B were mockups of 700 MW(e) designs, while 900 MW(e) designs were simulated in ZPPR-10C and -10D. The first three assemblies contained 19 CRPs, and in 10B seven of the positions were filled with mockup control rods. ZPPR-10D contained 31 CRPs, and by adding one and then six control rods and extra fuel, two additional critical variants of 10D were built prior to the end of the program. The critical fissile masses ranged from 1956 kg to 2740 kg and the core volumes (not including CRPs) ranged from 4590ℓ to 6158ℓ for the seven large homogeneous assemblies.

Some conclusions from the large core program are:

- Criticality: Using ENDF/B-IV data the ANL calculations of k_{eff} yield values within 0.1% of 0.984, and are within about 0.2% of results for smaller cores.
- Power Distributions: There is a tendency to overpredict all reaction rates in the outer radial regions by 2-3% relative to the core center.
- Reaction Rate Ratios: Relative to ^{239}Pu fission, ^{238}U capture is overpredicted by 7-9%, ^{235}U fission is overpredicted by 2-3%, and ^{238}U fission is underpredicted by 3-9%. These results are consistent with those from smaller cores.
- Control Worths: Average C/E values are close to those found for smaller homogeneous cores. However, consistent with the reaction

rates, there is a tendency to overpredict the worth in the outer positions relative to the center. The effects due to size, rod heterogeneity, geometry variations, B_4C enrichment, and rod interaction effects were well calculated using few-group diffusion theory.

- Sodium-void Reactivity: Calculations overpredicted positive reactivity worths of voiding central zones by about 17%, consistent with results from smaller cores.

B. HETEROGENEOUS FBR CRITICALS

There have been two ZPPR programs in support of the CRBR heterogeneous design. The first was completed in 1978, and the second is currently in progress. Basically, the critical configurations have consisted of a central blanket, 0-15 control rod positions, and fuel zones alternating with three internal blanket rings. For a typical configuration, the fuel zones have been enriched to 28% and have occupied a volume of about 2000 liters. The more detailed CRBR design mockups have included partially inserted control rods and fuel, blanket, and control pin zone. Measurements have included criticality, power distributions, reaction-rate ratios, control-rod worths, sodium-void reactivity, substitutions of thorium and plutonium in the blankets, and Doppler reactivity.

Important results from this program include:

- Many parameters were poorly calculated when simple methods were used, but when corrections were applied, the results were often consistent with those for homogeneous cores. Transport effects, plate streaming effects, coarse-mesh approximations, and geometric-modeling approximations were found to be relatively more important for the heterogeneous cores.

- Calculations of reactivity and reaction-rate distributions exhibit a consistent radial discrepancy, with results in the outer regions typically being overpredicted relative to near-central values by 5% and 2.5%, respectively.

- The well known difficulties in applying asymptotic cell processing methods at core-blanket interfaces were accentuated in the heterogeneous core calculations. This approximation caused a 12% underprediction of ^{238}U fission in the internal blanket regions.

- C/E ratios from both reference and model-corrected calculations of control worths were several percent lower than the equivalent homogeneous core results.

- Central sodium-void reactivities were overpredicted by the 15-20% - characteristic of the homogeneous core results.

C. CRITICALS FOR ALTERNATE CONCEPTS

Gas cooled fast breeder reactor (GCFR) critical experiments, planned jointly by ANL and the General Atomic Company, were conducted on ZPR-9 in

1975 and 1976. Program objectives were to provide clean tests of the basic nuclear data and analytical methods employed in GCFR design and safety analyses, and to provide experimental data on important safety coefficients; in particular, ^{238}U Doppler, steam entry, and He worths. The principal results of the GCFR Program were to confirm the general adequacy of computational methods and data in use for GCFR design, to indicate the importance of and provide methods for a proper treatment of neutron streaming in interpreting the ZPR mockup results, and to provide data against which attempts to sharpen the accuracy of steam-entry calculations -- which proved to be a difficult computational challenge -- could be tested.

Three advanced fuels compositions were studied during 1976 and 1977 as part of the Advanced Fuels Program on ZPR-9: a carbide composition characteristic of the outer core of a 1200 MWe design, a second carbide characteristic of an inner core composition, and an advanced oxide composition characteristic of a 1200 MWe advanced oxide design. The program was planned by ANL with participation by GE (Sunnyvale). The interest in advanced fuels stemed from its projected improvements in breeding properties over those of current mixed-oxide fuels. Measurements were made to assess the relative breeding and safety parameters. The most important result was the demonstration that the biases in the calculated predictions of the conversion ratio and of the safety coefficients do not change significantly from those in the current-oxide LMFBR systems.

Integral physics parameters of several representative, idealized meltdown LMFBR configurations were measured in mockup critical assemblies on the ZPR-9 reactor during late 1977.* The experiments were designed to provide results for the validation of basic nuclear data and calculational methods used in the neutronics part of LMFBR accident analyses. Large core distortions were introduced (involving 18.5% of the core volume), and the reactivity worths of configuration changes were determined as were power shapes and material worth profiles. The supporting analysis program was unique in that, for the first time, in addition to S_n and diffusion calculations, 3D explicit plate-by-plate continuous-energy Monte Carlo models were used to compute configuration eigenvalues -- this was an attempt to eliminate all but the basic nuclear data as a potential source of error in the eigenvalue calculation. The results of the program indicated that cross-section deficiencies in ENDF/B-IV exist which give rise, upon the massive-damage rearrangements considered, to a change of eigenvalue bias that exceeds a dollar in magnitude (though the individual biases are always less than 1% Δk). Except for a positive offset of ~0.5% Δk, transport theory, using cross sections which properly account for the heterogeneous nature of the unit cells, and at least a "transport correction" for anisotropic scattering, was found to be generally consistent with Monte Carlo for the cases considered and to share with Monte Carlo a configuration dependent bias when compared to experiment. Diffusion theory was not consistent with either experiment or Monte Carlo and predicted too large negative reactivity changes and too small positive reactivity changes for the configurations considered in the program.

*This program was sponsored by the Division of Reactor Safety Research U.S. Nuclear Regulatory Commission.

The Diagnostics Core Program, currently on ZPR-9, is directed at determining the causes of specific, long-standing, FBR physics discrepancies which impact the FBR design. A sequence of specially tailored cores is being studied to isolate the contributions made by nuclear data and by experimental and calculational techniques to these C/E discrepancies. Phase I of the program was directed at quantification of the contribution which unit cell heterogeneity and local experimental geometry make to the central worth discrepancy. The results placed a conservative upper bound of 5% on this value. Phase II of the program, currently in progress, is directed at isolating the contributions from basic data for the important isotopes of fast breeder compositions to the discrepancies on material worth, β_{eff} and c^{28}/f^{49}. The systematics of the variation of C/E with the fissile isotope (^{235}U, ^{239}Pu), the fertile-to-fissile atom ratio, and the hardness of the neutron spectrum (as controlled by the diluent/heavy-metal atom ratio) are being determined in a series of six assemblies.

IV. METHODS, CODES AND DATA

During the past decade there has been a significant increase in the power of computer hardware for scientific applications. The devlopment of methods, codes and data files which are amenable to the advanced computer systems and responsive to the need for greater rigor in reactor analysis has been an important priority in the U.S. FBR program.

The differential data base has evolved through five generations, culminating with the recent release of the ENDF/B-V files. These data are considerably more detailed than earlier files including increased attention to fission product, actinide, gamma-production, and cross-section correlation data. The improved accuracy of the files is a consequence of the strength of the differential measurements program as represented by the Fast Neutron Generator at ANL and the ORELA facility at ORNL, along with the use of improved evaluation methodologies.

The VIM Monte Carlo code was developed to provide a benchmark computational capability for the analysis of fast reactor systems based upon the ENDF/B data base. The point energy representation of cross-section data coupled with the generalized geometric detail provided by a combinatorial geometry capability make the VIM code a rigorous standard for neutron transport calculations.

The development of the MC^2-2/SDX code has provided a highly accurate data processing capability with the efficiency required for critical experiment and design analysis activities. An exhaustive validation study[2] was conducted to assess the accuracy of this processing capability for the analysis of ZPR criticals facilities. These studies demonstrated that the methods were accurately calculating the ZPR plate-type criticals provided neutron streaming is properly treated. New methods were developed by Gelbard which improved upon the classical methods of Benoist by extending the treatment to include lattices with planar voids, eliminating the non-uniqueness of the diffusion coefficient definition, and incorporating transport effects in the definition of the new diffusion coefficients. The Gelbard method was implemented using the VIM code to compute the anisotropic coefficients and has become a standard tool in the analysis of ZPR criticals facilities.

New code capability has made it possible to model the geometric complexity of the FBR with precision. The development of the DIF3D code provides a state-of-the-art computational capability for the solution of the multigroup finite difference diffusion equations in one-, two- and three-dimensional geometries. DIF3D uses optimized acceleration techniques which are specifically designed to treat fast reactor systems. A P_0 transport option in DIF3D has been used extensively in the ZPPR analysis of radial parfait core configurations. Using the same optimized iteration strageties along with an improved quasi-static approximation and a simplified thermal-hydraulics model, the FX2-TH code was developed to provide a two-dimensional space-time diffusion theory capability.

Approximate, fast-running methods have been developed to treat fast reactor depletion and dynamics problems. The SYN3D code, which is based upon a space-energy synthesis approximation to the finite-difference multigroup diffusion equations, has been used extensively with the REBUS-2 fast reactor depletion code to provide a three-dimensional depletion capability. Nodal methods have been developed which, by significantly reducing the number of unknowns, yield order-of-magnitude improvements in computing speeds with equivalent accuracy to finite difference methods for multigroup problems in orthogonal geometries.

Increased attention must be paid to the development of new methodologies which permit efficient design analysis of FBR systems. The work in nodal methods mentioned above could lead to an improved three-dimensional dynamics and depletion analysis capability. Extensions of generalized perturbation theory methods should permit the use of reference neutronics solutions to obtain high-order corrections as well as design oriented sensitivity effects. Synthetic acceleration methodologies hold the promise of allowing three-dimensional transport calculations with reasonable efficiencies. New methods work will be required in such areas as adjoint Monte Carlo, variance reduction methods and estimation theory to permit greater use of the Monte Carlo codes. Finally, methods work will be necessary to treat the non-asymptotic flux domain with sufficient rigor by data processing codes.

V. CONTEMPORARY STATUS AND TRENDS OF NUCLEAR DATA FOR FBR SYSTEMS

Uncertainties in basic nuclear data continue to be the largest cause of uncertainty in predictions of the physical performance of FBR systems. Differential data issues are characterized by both specific matters of precision and more general questions of physical concept. The status and recent trends of some of the most FBR-relevant basic-nuclear-data areas are outlined below.

Standard constants: The key quantities are $^{235}U(n,f)$ and ^{252}Cf(nubar) which, together, essentially govern the neutron-source term. $^{235}U(n,f)$ remains uncertain by 2-3%, implying larger uncertainties in all other fission cross sections. ^{252}Cf(nubar) has recently varied by 0.5-1.0% and with it essentially all actinide nubar values. The trend in $^{235}U(n,f)$ is downward but partly compensated for by a trend toward increased ^{252}Cf(nubar) values. In order to significantly improve the situation several independent measurements to accuracies of 0.5-1.0% in the case of $^{235}U(nf)$ and 0.2-0.3% for ^{252}Cf(nubar) are needed.

Actinide nuclides: Recent experimental results indicate increased neutron total cross sections in the unresolved resonance region, in some cases (e.g., ^{238}U) by 5-10%. Recent measurements of neutron inelastic scattering from fertile nuclides have led to generally larger cross-section values while at the same time changing the energy transfer matrix in a manner that makes the larger cross sections more acceptable in integral calculations. It remains very difficult to measure fissile-nuclide inelastic cross sections and, consequently, the results are uncertain. However, fissile-nuclide elastic-scattering cross sections are now determined with good accuracies that suggest lower inelastic scattering cross sections relative to those generally employed in FBR calculations. Fission cross sections of prominent actinides are known relative to those of ^{235}U to 1-2% accuracies, thus the ^{235}U reference standard remains the governing factor. Fast-neutron capture cross sections of fertile nuclides are known to 5-8% and, in the case of ^{238}U, remain larger than suggested by integral studies. Nubar uncertainties are dominated by those of the reference standard, as noted above. The most recent evaluations have generally increased the average fission-neutron energies although there are large uncertainties in these data. Knowledge of delayed-neutron emission from specific precursors is much improved but the average delayed-neutron yields have not markedly changed. Recent measurements have provided new transplutonium data, notably neutron total and fission cross sections, and the corresponding evaluated data sets are improving. The latter make extensive use of theoretical extrapolations that probably are sufficiently accurate for most FBR use.

Structure and coolant nuclides: In this area the problems involve the measured data, the evaluation and the calculational usage. Uncertainties are reflected in the wide variation between group-transport cross sections as derived in various national programs.[3] Experimental knowledge of resonance cross sections has very much improved particularly with respect to radiative capture. This improvement is not necessarily reflected in the evaluated files due to physically inappropriate representations. The quality of the basic data in the unresolved resonance region is variable. Measurements have often emphasized resolution to the detriment of absolute magnitude and self-shielding effects have frequently been ignored with consequent distortion of 10% or more in some cases. Inelastic neutron-scattering cross sections in this region are generally large and measurements are now providing accuracies approaching 5%.

Fission products: A data base of wide scope is available. However, for applications sensitive to fission-product data, there is concern for data quality. The fission products are neutron-rich nuclides characterized by inversion of the shell structure and/or large static deformations. They are usually very radioactive and thus the data are largely obtained by theoretical extrapolation from measured values in neighboring regions. This procedure generally uses "global" models that can be far from reality in specific cases. Inelastic-scattering cross sections of these nuclides are among the largest known and not well represented by evaluation. This shortcoming is probably of minor note but fast-neutron capture is a more serious concern and discrepancies of factors of 2-5 in fission-product capture cross sections are known to exist and are supported by fragmentary integral results. It is not reasonable to soon expect large improvements in the measurement of active fission-product cross sections. It is reasonable to look forward to better knowledge of the stable cases with improved theoretical extrapolation to the active products.

VI. INTERNATIONAL COMPARISONS

Some indication of the progress and status of FBR design calculations can be derived from results of international comparison calculations.[3-6] Such studies attempt to evaluate and document the agreement and differences in the calculations of key FBR physics and safety parameters as a function of the different nuclear data sets and processing codes used by the participants. The most recent of these studies, the 1978 NEACRP/IAEA international comparison calculation of a large sodium-cooled fast breeder reactor is reviewed first. Sixteen solutions were contributed from the ten countries participating in the comparison. These solutions represented twelve different basic nuclear data sets; six of these were evaluated data sets (i.e., non-adjusted data sets) and six were adjusted. Four of the solutions were based on ENDF/B-IV data. Some of the major conclusions were:

- Participants who had obtained good agreement between their calculations and measurements of fission rate and material worth distributions on 300 MWe size critical experiments exhibited large differences (up to 20%) on the calculations of these distributions in the 1250 MWe size comparison calculation.
- There were large variations (over 45%) in the calculated central control rod worth.
- The 1σ variation in the calculated eigenvalues for the reference configuration was 0.013 δk.
- The 1σ variation for the core breeding ratio (4.3%) was about the same as that for $^{28}c/^{49}f$ (3.7%).*
- Strong correlations were observed among the parameters k_{eff}, k_∞, breeding ratio, $\langle\sigma_c^{28}\rangle$, and $^{28}c/^{49}f$ (viz., solutions having high $\langle\sigma^{28}\rangle$ and $^{28}c/^{49}f$ values had high breeding ratio and low k_{eff}).*
- The 1σ variations were ~12-17% in the sodium-void reactivity calculations and were ~11-14% in the fuel Doppler worth calculations.

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*c and f denote per-atom capture and fission rates and $\langle\sigma\rangle$ denotes a spectrum-averaged effective cross section.

- Significant differences were observed between the one-group effective cross sections from the adjusted data sets compared to those from the non-adjusted data sets. Furthermore, the scatter in the values of the adjusted cross sections was about as large as the scatter in the values of the non-adjusted cross sections, and the scatter due to different processing of the same data set was about as large as the scatter between data sets.

- The scatter in the integral parameters calculated from adjusted data sets was reduced (by a factor of 2-3) for k_{eff}, k_{∞}, breeding ratio and $^{28}c/^{49}f$. On the other hand, the scatter in $^{28}f/^{49}f$, central material worths, and safety parameters was about as large among the users of adjusted data sets as it was among users of non-adjusted data sets.

The variation in the results observed in the NEACRP comparison is a measure of the differences in the nuclear-data files and/or the data-processing codes in the participating countries, but the actual design uncertainties are smaller due to the application of experimentally derived bias factors.

Finally, it is interesting to compare these conclusions with the results of the "Baker Model"[4] comparison of 1970. In that study, the 1σ variation among the calculations for the reference case was 0.012 for the eigenvalue and 0.041 for the internal breeding gain. The similarity between these variations and those observed in the 1978 comparison is not be unexpected, however, since the comparisons include results of both adjusted and non-adjusted data sets and therefore a spread in results is expected.

VII. CRITICALITY

Predictions of criticality may be checked directly with the few fast power reactors plus a comparatively larger number of critical assembly measurements. For eigenvalue calculations using ENDF/B-IV data, of conventional LMFBR type assemblies in the ZPR-6, ZPR-9, and ZPPR critical facilities (which includes about a dozen major assemblies):

- The range of C/E values is only from 0.983 to 0.986 (or 0.3% δk) with a mean value ±1σ of 0.9843 ± 0.0010.

Therefore, although the bias on eigenvalues calculated with ENDF/B-IV is ~1.5%, the designer can account for the bias with good precision (~0.15%).

The critical-mass measurement is perhaps the most simple and direct integral measurement obtained from critical assemblies. It involves nothing more than a complete specification of the contents and conditions of the "as-built" configuration. All of the specifications are known accurately, and therefore the measurement of the critical mass is one of the most precise experimental values obtained.

- The experimental uncertainty (1σ) in the measured k_{eff} of the ANL-ZPR fast critical assemblies is typically ±0.05 - 0.10% δk.

- The principal components of this uncertainty are due to uncertainties in the isotopic composition and corrections associated with the interface of the two assembly halves.

The post-analysis of a ZPR fast critical assembly determines a calculated reactivity and C/E bias on k_{eff} for the "as-built" configuration (as opposed to calculating the critical mass of the k = 1.0 system). In principle, this calculation is as simple and direct as the measurement. However, the uncertainties in the cross-section processing, methods, and modeling approximations alone (i.e., neglecting uncertainties in the basic nuclear data files) produce

- a 1σ uncertainty of $\pm 0.2 - 0.5\%$ δk in the calculated k_{eff}.

Within this range of methods uncertainties the smaller values are appropriate for calculations of conventional LMFBR type assemblies; the larger values appropriate for non-conventional LMFBR assemblies, such as heterogeneous designs or distorted meltdown configurations. The validity or suitability of the basic nuclear data to predict critical mass is tested by comparing the overall bias in eigenvalue C/E to these experimental and calculational method uncertainties.

One might consider C/E values of k_{eff} for typical LMFBR-type critical assemblies obtained with ENDF/B Versions III, IV and V as indicative of the progress or status of predicting criticality over the past decade using ENDF/B nuclear data. Using ZPR-6 Assemblies 6A and 7 for illustration:

Eigenvalue C/Es Using Different Versions of ENDF/B

ZPR/Assembly	Fuel Type	Version III	Version IV	Version V
ZPR-6/6A	^{235}U	0.986	0.984	0.988
ZPR-6/7	^{239}Pu	0.985	0.983	0.996

It may be noted that a k_{eff} bias of ~1.5% δk (underprediction) was typical with ENDF/B Versions III and IV. Although limited data testing results are available with Version V data, it appears that Version V will largely eliminate the C/E bias for the dilute ^{239}Pu-fueled systems, with less effect on the ^{235}U-fueled systems, thereby creating a C/E bias between the fuel types.

Extrapolation to alternative designs (e.g., heterogeneous systems, accident configurations, GCFR designs) would similarly utilize bias factors obtained from analysis of critical assembly measurements, would result in much higher uncertainties (perhaps $\pm$ 0.5 - 0.7% δk) due to the limited critical assembly data base. Additional work is required in these areas.

VIII. SODIUM VOID EFFECT

Sodium coefficients have been an issue in LMFBR design and safety analysis for two decades. The emergence of heterogeneous cores as the consensus approach in the U.S. indicates that the issue is still extant.

Much progress has been made in the last decade in understanding the physics aspects of sodium-void reactivity. This has resulted from two advances: (1) significant progress in both computer capability and code and methods development, and (2) the acquisition of a large amount of critical experiment data.

In order to assess the accuracies achievable, new calculations have been done of many past sodium void experiments. These calculations used as a reference method three-dimensional diffusion calculations (exact perturbation theory applied), in 28 groups with ENDF/B Version IV data. Neutron streaming effects were accounted for by use of the Benoist prescription. The experimental data were re-evaluated using a consistent approach, and ENDF/B Version V delayed neutron yields were used for the calculated reactivity scale.

Data from ZPR-6 Assembly 7 and ZPPR Assemblies 3, 5, 7, 9, and 10 were included. As outlined earlier, these assemblies represent a range of reactor sizes, arrangements, and material distributions. For 45 different inner core experiments, including voiding of both central and off-center zones, the calculations (C) and experimental values (E) compared with an average C/E ratio of 1.13 with a standard deviation of 0.09 for the distribution. For voiding in axial blankets, analysis of eight experiments resulted in $\overline{C/E} = 1.05 \pm 0.14$.

The dispersion in $\overline{C/E}$ was further reduced by applying corrections for neutron transport and finite mesh size. This was done for a central core zone from each of eight assemblies; results are shown below. The corrections had little effect except in the heterogeneous core and in those cases where the voided zone surrounded a control rod singularity. The average C/E when corrected calculations are used is 1.19 ± 0.05, where the uncertainty in the measured values is ±1-2%.

Central Zone Sodium-Void Results from Eight Critical Assemblies

Assembly	Zone Volume, ℓ	Comment	Calculated Worth, (Reference Method) ¢/kg	C/E Ratios: Reference Calculation	C/E Ratios: Corrected Calculation
ZPR-6/7	8.4	Clean Benchmark	2.59	1.230	1.230
ZPPR-3/3	195	Central CRP[a]	1.15	1.106	---
ZPPR-5A	234	Central CRP	0.84	1.208	1.224
ZPPR-5B	234	Central CR	1.33	1.151	1.224
ZPPR-7A	305	Het. Core Benchmark	0.868	1.073[b]	1.162
ZPPR-9	121	Large Clean Benchmark	1.028	1.095	1.107
ZPPR-10A	214	Central CRP	0.913	1.158	1.169
ZPPR-10B	214	Central CR	0.736	1.105	1.200
			Average C/E	1.14 ± 0.06	1.19 ± 0.04

[a]Control rod position.

[b]Special cross section procedures for ZPPR-7A; infinite medium of coupled core-blanket cells.

These results for the dispersion in C/E ratios appear to agree roughly with those reported by Butland et al.,[7] who analyzed a substantial number of sodium void experiments performed at the British ZEBRA facility.

As has been done in (at least) the U.K., the U.S., and France, a least-squares fitting procedure was used to develop separate bias factors for the calculated leakage and non-leakage terms. It was found that for the 45 inner-core results, these factors were 0.88 and 0.87, respectively, to be applied to the reference calculations. Since the biases were nearly equal, little improvement in C/E dispersion resulted--the average went from 1.13 ± 0.09 to 0.99 ± 0.08.

Further sodium-void experiments will be done to fill in at least two deficiencies in the current data. First are experiments in larger heterogeneous cores. There is no evidence to suggest that major changes will be observed, but confirmation would be useful. Second, experiments need to be repeated in more homogeneous unit cells than the usual drawers filled with plates. The ZPR materials inventory now includes fuel pins of $(U\text{-}Pu)O_2$ and depleted UO_2 radial blanket pins. While two sodium void experiments with core pins have been done before, results were contradictory and need to be carefully repeated. The most recent results (ZPPR-5) and the results of pin experiments done in the U.K., both indicate that similar C/E ratios are obtained with either pin or plate construction.

IX. CONTROL ROD WORTHS

Reductions in uncertainties associated with control rod effects in FBRs have resulted from successful research efforts over the last ten years. In his 1972 summary, Avery[1] characterized improved knowledge of control effects as one of the more pressing needs of the reactor designer, as neither experimental nor calculational methods had been verified. Since then, literally hundreds of control worth measurements have been included in the ZPPR programs. Calculational methods for application to reactor design have been validated through analysis of the experiments. For designs that have been studied in the ZPPR experiments, uncertainties in control worth C/Es have been reduced to less than 5%. The component of these uncertainties due to current measurements is only about 2%.

The ZPPR experimental data base for control rod effects includes:

- worths of single rods and rod banks,
- rod reactivity interaction effects,
- maximum worth of singly and doubly faulted rod conditions,
- power distributions in and around rods,
- the effect of inserted control rods on k_{eff} predictions,
- the effects of rod size, geometry, heterogeneity, and B_4C enrichment,

- the worth of Ta and Eu_2O_3 relative to B_4C rods,
- the variation of rod worth with axial insertion and
- the effects of sodium-filled control rod positions.

Calculations using typical design methods -- coarse-mesh, few-group, two-dimensional diffusion-theory calculations -- produce C/E ratios with a total range of 0.9 to 1.1, a mean value of about 1.04, and a standard deviation for the distribution of about 3.5%. Differences of up to 15% occur between C/E values for heterogeneous and homogeneous cores, while spatial variations of 5-10% are seen within the heterogeneous and large-homogeneous core results. Large interaction effects of 50-60% were also observed in heterogeneous and large cores, but were predicted to within 1-3% by design methods. Simple calculations predicted the effects of enrichment and geometry changes to within 1-2%.

Accounting for modeling approximations has reduced the spread in C/E values to 13%, with good improvement in some spatial variations. Corrected C/E's tend toward 1.1, consistent with the central worth discrepancy.

Until efforts to resolve the discrepancies are more successful, extrapolation of current methods to different reactor designs will introduce large uncertainties. Plans for future work include development of methods to handle axial streaming in control positions, consistent application of refined calculational methods, studies of sensitivities to cross-section uncertainties, and measurements in large heterogeneous cores.

X. REACTION RATE DISTRIBUTIONS AND RATIOS

Spatial distributions and absolute ratios of $^{239}Pu(n,f)$, $^{235}U(n,f)$, $^{238}U(n,f)$, and $^{238}U(n,\gamma)$ reaction rates have been key measurements in the critical facilities. Thin foils, placed between plates in the ZPR drawers, are irradiated and subsequently analyzed by high-resolution gamma-ray spectroscopy of fission or capture products. The data have statistical uncertainties of 1% or less in the core zones. Factors generated in auxilliary experiments convert the foil data to cell-averaged reaction rates and ratios with estimated uncertainties of up to 2-3%, depending on the reaction. Special experiments, including intercomparisons with other laboratories, are used to verify the foil-counting techniques.

There have been several advances in methods used to calculate reaction rates. Improvements have been made in the MC^2-2/SDX codes for cell processing and have been validated by comparisons with the VIM Monte Carlo code. More use has been made of three-dimensional diffusion and two-dimensional transport calculations. The treatment of plate-cell streaming with directional diffusion coefficients has become routine in ZPR calculations.

Calculations of reaction rates from all the ZPPR assemblies have identified some common problems with axial distributions. Reaction rates near the top and bottom of the core are underpredicted by about 1%, with the effect being accentuated for the threshold reaction $^{238}U(n,f)$. Reaction rates are also underpredicted deep within the axial blankets, and the discrepancy increases with penetration into the blanket.

No systematic discrepancies in calculated radial power distributions can be clearly identified for the smaller homogeneous cores. Few-group diffusion-theory calculations lead to bias factors of unity with standard deviations for the C/E distributions of about 2%. Maximum deviations from measurement are about 5% and occur at core/blanket boundaries or adjacent to control rods.

The heterogeneous-core results displayed a systematic radial bias in calculated fission rates. Values in the outer fuel rings were over-predicted relative to the inner fuel rings by 2-5%, varying somewhat with core arrangement, scan direction, and calculational method. The discrepancy diminished when plutonium was added to the internal blankets to simulate equilibrium conditions. Variations in $^{238}U(n,f)$ of up to 100% between adjacent fuel and blanket assemblies were mispredicted by as much as 25% using 28-group diffusion-theory methods. Both special cross-section processing (appropriate to the coupled cells) and transport calculations were required to reduce the C/E discrepancy to 2%.

Results from the large ZPPR cores are of considerable interest in view of the significant discrepancies among power distributions calculated by participants in the most recent NEACRP benchmark comparison.[3] Calculations of ZPPR-9 and -10 with ENDF/B-IV data were found to overpredict reaction rates in outer regions relative to the core center, but only by 2-3%. Cross section sensitivities for these ZPPR cores are about 70% of those for the 1250 MWe NEACRP core. Discrepancies could be bigger as neutronic coupling is reduced in the larger or heterogeneous cores. Additional experimentation is necessary to determine the magnitude of the problem.

The C/E values for reaction-rate ratios from calculations using ENDF/B-IV data do not vary widely within the range of ZPPR cores. Relative to ^{239}Pu fission, ^{238}U capture is overpredicted by 5-9%, ^{238}U fission is underpredicted by 3-9%, and ^{235}U fission is overpredicted 1-3%. Of these the first is the most important, because errors in $^{238}U(n,\gamma)$ predictions directly impact computed breeding performance and burnup reactivity changes. This is one of the most serious problems in U.S. FBR physics.

XI. MATERIAL WORTH MEASUREMENTS

The following table summarizes the range of values of (C/E) for fissile material worth and worth ratio experiments made at core center in the ZPPR and ZPR critical assemblies. The ranges shown, which are based on ENDF/B-IV data, summarize many measurements and apply to both conventional breeder simulations as well as alternative breeder compositions and configurations. These ranges are not typical of C/E results based on the adjusted data sets (e.g., FGL5) where the values are closer to unity.

C/E on Small Sample Worth Ratios and Fissile Worth

ρ^{49}	$\rho^{10}B/\rho^{49}$	ρ^{28}/ρ^{49}	ρ^{SS}/ρ^{49}
1.08 - 1.30	0.85 - 0.95	0.87 - 1.04	1.05 - 1.22

The measurements are made by introducing a small material sample ($\lesssim$30 gm) into the core using a low frequency oscillator mechanism and inferring the reactivity in ¢ or Ih units from the resulting time variation of neutron detectors via the point kinetics equations. The calculations are made in ~30 group RZ or XY diffusion first order perturbation theory. The calculated worth in Δk units is converted to ¢ or Ih units using calculated values of β and effective delayed neutron lifetimes.

The ~15% overprediction of the fissile worth itself -- which was discussed in Avery's paper[1] (the "central worth discrepancy") -- remains despite the intervening 10 years of effort and two revisions of ENDF. The discrepancy is, in fact, global rather than central -- it is typically found that the calculation adequately predicts the spatial shape of the fissile worth once the amplitude is normalized to the central value.

As pointed out in Avery's paper, the existence of the worth discrepancy casts a nagging ambiguity over comparisons of measured and calculated worths of all kinds and necessitates caution in the application of worth bias factors to kinetics predictions in design work. The discrepancy has been studied for some time, but remains unresolved. Recent results indicate:

- Checks on the calculational accuracy of β_{eff} and effective delayed neutron lifetime indicate that a miscalculation of the reactivity scale conversion factor is responsible for about 3 ± 3% of the ~15% central worth discrepancy, and can explain neither the entire discrepancy nor the differences in C/E on worth ratio between control, structural, and fertile materials relative to fissile. A ~4% difference between β_{eff} C/E values in plutonium and uranium cores tends to explain a bias in fissile worth C/Es for these cores.

- Detailed experimental investigations have shown that sample self-shielding, flux distortions, and kinetics data reduction can be responsible for at most 6% uncertainty relative to a first order perturbation theory calculation.

- Theoretical and numerical studies have shown that the failure to use bilinear weighting for cross section preparation contributes insignificant errors to calculated fissile and control material worths -- though the induced errors are nontrivial for fertile and are large for structural materials.

The potential contribution made by basic nuclear data exclusive of the delayed neutron data is currently under systematic study in the Diagnostics Cores Program on ZPR-9.

XII. CONCLUSIONS

Data are available now to assess the predictability of several different core concepts relative to a reference Demo-sized homogeneous design. The results indicate that large homogeneous cores can be predicted nearly as accurately as the reference design but that Demo-sized heterogeneous cores require more sophisticated analytical methods to achieve comparable accuracies. Large heterogeneous cores are likely to be even more challenging. Predictions of the carbide benchmark core revealed no serious problems; however, the GCFR cores required special treatment for neutron streaming. Comparable accuracies were not achieved for the meltdown configuration even when the most sophisticated methods were used indicating possible data deficiencies in the energy ranges of interest for these cores.

Considerable progress has been achieved in both the understanding and prediction of sodium-void reactivities and control rod worths. Although there is an absolute bias of about 1.5% δk using ENDF/B-IV data, criticality predictions of LMFBR configurations are very consistent. Radial power shapes are well-predicted in homogeneous Demo-sized cores but small shape discrepancies have appeared for heterogeneous and large homogeneous cores. Reaction rate ratio measurements have yielded consistent C/E values across the range of core designs. The 5-9% discrepancy in the prediction of the $^{238}U(n,\gamma)/^{239}Pu(n,f)$ ratio still exists resulting in a large uncertainty in the prediction of both breeding performance and burnup reactivity changes. This is an important problem requiring further investigation.

Progress on Doppler in the past few years has been limited and, as such, Doppler was not covered in this paper. The small-sample central worth discrepancy still exists but a consistency among the small sample, sodium void and control rod worth C/Es seems to be emerging. The values are all falling in the range of slightly less than 1.1 to 1.2. The central worth discrepancy is a serious concern, however, and should be resolved.

Internationally, the predictions of FBR parameters by the various national data sets do not appear to be converging but this is not unexpected due to the widespread use of both adjusted and non-adjusted sets. An international comparison of biased predictions would be useful.

The physics data from FFTF will begin to become available soon. There is interest at ANL in how well the physics parameters were predicted from the critical experiments data, and, in the longer run, the available data on fuel cycle neutronics. Also it appears now that heterogeneous designs will continue to be emphasized in the U.S. and that this will have a significant impact on the physics programs. More rigorous computational methods must be developed and critical experiments for large heterogeneous cores are required.

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